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**High-Temperature Gas-Cooled
Reactor Safety Studies for the
Division of Accident Evaluation
Quarterly Progress Report,
July 1–September 30, 1984**

S. J. Ball
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C. F. Weber
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Prepared for the U.S. Nuclear Regulatory Commission
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HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR
THE DIVISION OF ACCIDENT EVALUATION QUARTERLY
PROGRESS REPORT, JULY 1-SEPTEMBER 30, 1984

S. J. Ball, Manager
J. C. Cleveland
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J. H. Wilson

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PRIOR HTGR SAFETY REPORTS

Quarterly Progress Reports

<u>Ending date</u>	<u>Designation</u>
September 30, 1974	ORNL/TM-4798
December 31, 1974	ORNL/TM-4805, Vol. IV
March 31, 1975	ORNL/TM-4914, Vol. IV
June 30, 1975	ORNL/TM-5021, Vol. IV
September 30, 1975	ORNL/TM-5128
December 31, 1975	ORNL/TM-5255
March 31, 1976	ORNL/NUREG/TM-13
June 30, 1976	ORNL/NUREG/TM-43
September 30, 1976	ORNL/NUREG/TM-66
December 31, 1976	ORNL/NUREG/TM-96
March 31, 1977	ORNL/NUREG/TM-115
June 30, 1977	ORNL/NUREG/TM-138
September 30, 1977	ORNL/NUREG/TM-164
December 31, 1977	ORNL/NUREG/TM-195
March 31, 1978	ORNL/NUREG/TM-221
June 30, 1978	ORNL/NUREG/TM-233
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December 31, 1978	ORNL/NUREG/TM-314
March 31, 1979	ORNL/NUREG/TM-336
June 30, 1979	ORNL/NUREG/TM-356
September 30, 1979	ORNL/NUREG/TM-366
December 31, 1979	ORNL/NUREG/TM-383
March 31, 1980	ORNL/NUREG/TM-397
June 30, 1980	ORNL/NUREG/TM-415
September 30, 1980	ORNL/NUREG/TM-429
December 31, 1980	ORNL/TM-7809
March 31, 1981	ORNL/TM-7889
June 30, 1981	ORNL/TM-8091
September 30, 1981	ORNL/TM-8128
December 31, 1981	ORNL/TM-8260
March 31, 1982	ORNL/TM-8443/V1
June 30, 1982	ORNL/TM-8443/V2
September 30, 1982	ORNL/TM-8443/V3
December 31, 1982	ORNL/TM-8443/V4
March 31, 1983	ORNL/TM-8921/V1
June 30, 1983	ORNL/TM-8921/V2
September 30, 1983	ORNL/TM-8921/V3
December 31, 1983	ORNL/TM-8921/V4
March 31, 1984	ORNL/TM-9267/V1
June 30, 1984	ORNL/TM-9267/V2

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Topical Reports

S. J. Ball, *ORECA-I: A Digital Computer Code for Simulating the Dynamics of HTGR Cores for Emergency Cooling Analyses*, ORNL/TM-5159 (April 1976).

T. W. Kerlin, *HTGR Steam Generator Modeling*, ORNL/NUREG/TM-16 (July 1976).

R. A. Hedrick and J. C. Cleveland, *BLAST: A Digital Computer Program for the Dynamic Simulation of the High Temperature Gas Cooled Reactor Reheater-Steam Generator Module*, ORNL/NUREG/TM-38 (August 1976).

J. C. Cleveland, *CORTAP: A Coupled Neutron Kinetics-Heat Transfer Digital Computer Program for the Dynamic Simulation of the High Temperature Gas Cooled Reactor Core*, ORNL/NUREG/TM-39 (January 1977).

J. C. Cleveland et al., *ORTAP: A Nuclear Steam Supply System Simulation for the Dynamic Analysis of High Temperature Gas Cooled Reactor Transients*, ORNL/NUREG/TM-78 (September 1977).

S. J. Ball et al., *Evaluation of the General Atomic Codes TAP and RECA for HTGR Accident Analyses*, ORNL/NUREG/TM-178 (May 1978).

J. C. Conklin, *ORTURB: A Digital Computer Code to Determine the Dynamic Response of the Fort St. Vrain Reactor Steam Turbines*, ORNL/NUREG/TM-399 (March 1981).

S. J. Ball et al., *Summary of ORNL Work on NRC-Sponsored HTGR Safety Research, July 1974-September 1980*, ORNL/TM-8073 (March 1982).

FOREWORD

High-temperature gas-cooled reactor safety studies at Oak Ridge National Laboratory are sponsored by the Division of Accident Evaluation (formerly the Division of Reactor Safety Research), which is part of the Office of Nuclear Regulatory Research of the Nuclear Regulatory Commission.

This report covers work performed from July 1–September 30, 1984. Previous quarterly reports and topical reports published to date are listed on pages v and vi. Copies of the reports are available from the Technical Information Center, U.S. Department of Energy, Oak Ridge, TN 37831.

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR
THE DIVISION OF ACCIDENT EVALUATION QUARTERLY
PROGRESS REPORT, JULY 1-SEPTEMBER 30, 1984

S. J. Ball, Manager

J. C. Cleveland C. F. Weber
R. M. Harrington J. H. Wilson

ABSTRACT

Modeling and code development work on the modular High-Temperature Gas-Cooled Reactor (HTGR) were continued. The longer-term heatup accident scenario in which cavity wall cooling is lost was also modeled. Sensitivity studies were run for a variety of parameter variations. Fission-product (FP) release and transport experiments were completed for several additional elements.

1. HTGR SYSTEMS AND SAFETY ANALYSIS

Work for the Division of Accident Evaluation (formerly Reactor Safety Research) under the High-Temperature Gas-Cooled Reactor (HTGR) Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during this quarter included continuation of model and code development for the modular HTGRs and for fission-product (FP) redistribution during severe accidents in large HTGRs. FP release and transport experiments and technical assistance work on a Fort St. Vrain (FSV) technical specification (tech spec) review were continued.

1.1 Modular HTGR Simulation Development

R. M. Harrington
S. J. Ball J. C. Cleveland

1.1.1 Introduction

The most recent quarterly report of this series¹ describes the development of a dynamic simulator for the 250-MW modular HTGR and reports the results calculated for several different heatup accidents. This quarter's work includes explorations of the effect of various parametric and modeling assumptions on the peak fuel temperatures following a reactor scram with the following concomitant casualties: loss of forced circulation of primary coolant, depressurization of the reactor vessel, and loss of cooling water flow to the steam generators. For these sensitivity calculations, described below in Sects. 1.1.2-1.1.6, it is assumed that the reactor vessel cavity cooling system is in operation throughout the ~30-h period between accident initiation and the peak fuel temperature. (This is a reasonable assumption since the

cavity cooling system, which is normally active, is designed with the capability for passive operation relying on the natural circulation of cooling water and is provided with a 2-week supply of cooling water.) The core and loop models (Fig. 1), as described in Ref. 1, were used with little or no modification; therefore, the accompanying detailed description of the model will not be repeated here.

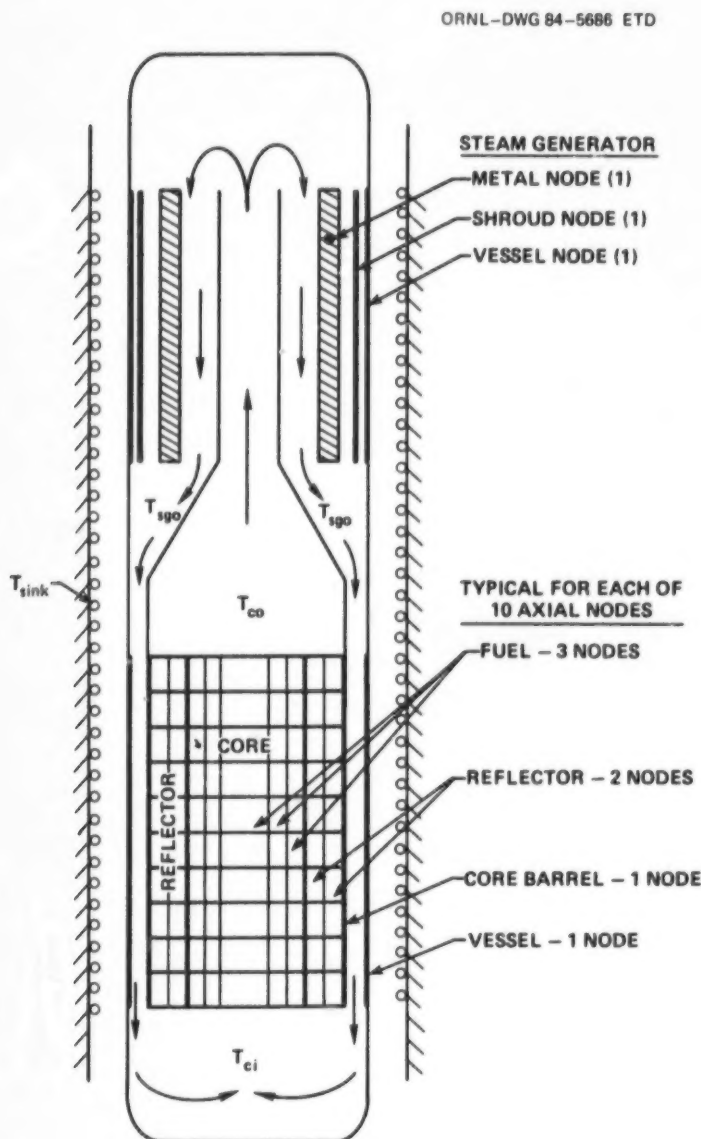


Fig. 1. Nodal structure of modular HTGR thermal model.

In addition to the three major casualties noted above, for the depressurized heatup it is possible to postulate that the reactor vessel cavity cooling system (both its active and passive modes) might fail. This failure might be coincident with accident initiation, or it could be assumed to happen after the 2-week supply of liner cooling water runs out. For such an accident it is of interest to know if fuel or major structural components, such as the reactor vessel, would be damaged by excessive temperature excursions during the days or months following the loss. Section 1.1.7 describes a long-term simulation model developed to address these questions and presents typical results. This long-term simulation includes a detailed model of heat transfer to the earth that surrounds the ~10-m-diam by 30-m-high reactor cavity.

1.1.2 Effect of decreased core power density

The input to the modular HTGR dynamic simulator program was modified, as appropriate, to model the following recent changes in reference parameters: a decrease in core power density from 4.1 W/cm^3 to 3.7 W/cm^3 (but total core power maintained at 250 MW), a 22% increase in normal core flow, and an increase in normal core inlet temperature from 220°C to 255°C . The decrease in core power density was achieved by increasing both the core pebble bed height and the core diameter. With the higher core flow, the average full power coolant temperature at core outlet is 689°C , or $\sim 60^\circ\text{C}$ lower than with the previous core flow and inlet temperature.

The decreased power density and increased core flow combine to decrease both the initial steady state temperatures and the transient peak in maximum core temperature. The initial, full power, steady state, average fuel temperature is reduced by 11°C , and the corresponding maximum (hot-node) initial fuel temperature is reduced by 88°C . The transient peak hot-node fuel temperature during the depressurized heatup is reduced from 1728°C at 1700 min^1 to 1699°C at 1842 min ; however, the transient peak in average fuel temperature is increased by $\sim 6^\circ\text{C}$. Further sensitivity studies will be required to determine the reason(s) for the lack of improvement of average fuel temperature and for the smaller than expected improvement in the transient peak hot-node temperature.

Figure 2 shows the depressurized heatup with the new core power density, flow, and inlet temperature. The corresponding transient response for the previously used values of core flow, inlet temperature, and power density is shown in Fig. 2 of Ref. 1.

1.1.3 Effect of GAT input for fuel specific heat, conductivity, and decay heat

The previously reported¹ Oak Ridge National Laboratory (ORNL) calculation of a 1728°C transient peak in maximum fuel temperature during depressurized heatup is well in excess of that predicted by General Atomic Technologies, Inc., (GAT) and Interatom studies,² which both predict peak fuel temperature below the 1600°C design goal for the 250-MW HTGR. To see where the difference might lie, input was modified,

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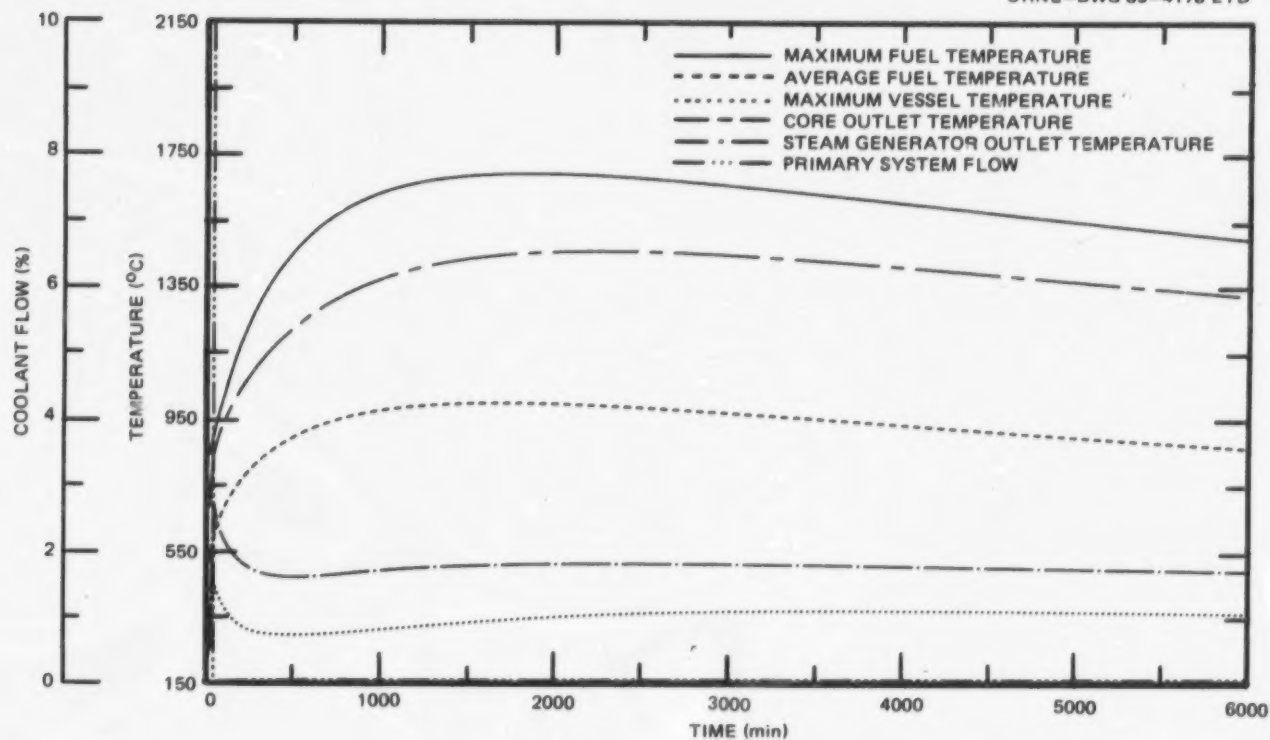


Fig. 2. Loss of forced circulation with depressurization and loss of feedwater flow.

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and runs were performed using data for certain key parameters, as specified in GAT Report No. 906792.³

Figure 3 shows the ORNL and GAT input for fuel specific heat. For temperatures below 1400°C, they are identical. As temperature increases above 1400°C, the ORNL specific heat is not allowed to increase, whereas the GAT curve increases rapidly. The ORNL analysts did not allow the rapid increase above 1400°C because they are familiar with other data⁴ that show that fuel specific heat tends to approach an asymptotic value as temperature increases through this range (Fig. 3, lowest curve). It was concluded that the correlation referenced by GAT does not adequately represent the specific heat for a fuel temperature range above 1400°C. The effect of this difference on peak fuel temperature is small: substitution of the complete GAT curve into the input for the ORNL simulation resulted in only a 9°C reduction of predicted peak fuel temperature.

Figure 4 shows the ORNL and GAT correlations for fuel pebble bed conductivity, expressed here for conductive and radiative heat exchange across the pebble bed (for which the thermal resistance of the interiors of the pebbles is not limiting). The linear relationship utilized by ORNL⁵ shows a significantly higher conductivity at the lower end of the temperature scale but an ~10% lower conductivity at the higher end of

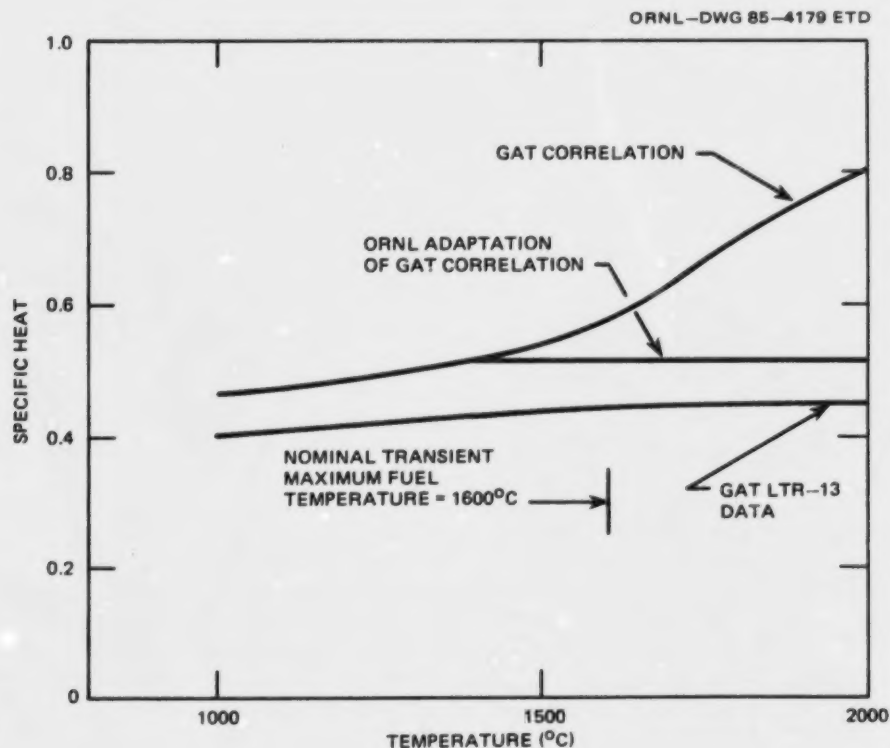


Fig. 3. Specific heat of pebble bed fuel.

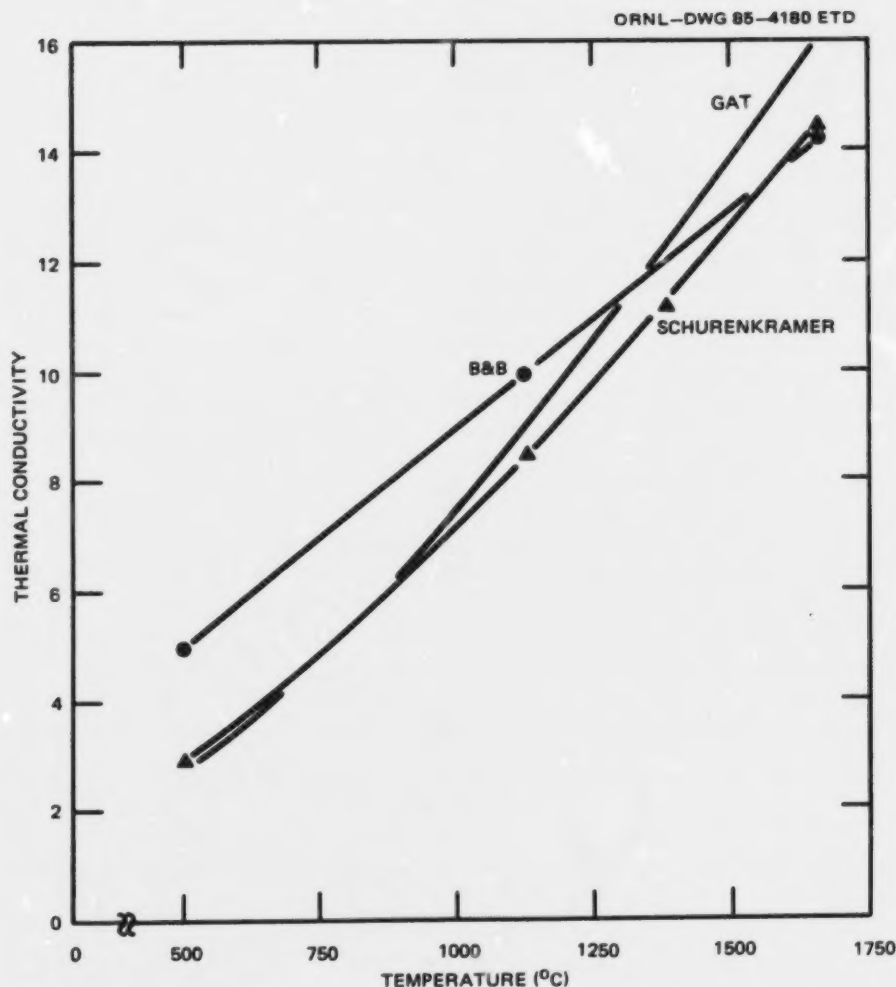


Fig. 4. Thermal conductivity of pebble bed core.

the temperature scale. Substitution of the GAT curve into the ORNL model reduced the calculated peak temperature by 51°C.

Figure 5 shows both the correlation cited by GAT analysts in Ref. 3 for decay heat following scram from full power and the correlation used by ORNL analysts in the modular HTGR simulation. The correlation used by ORNL, based upon a report published by Kernforschungsanlage Jülich (KFA),⁶ is very similar to the 1973 American Nuclear Society (ANS)-recommended standard relationship, with effectively infinite irradiation but with no allowance for the buildup of actinides or other activation products. Substitution of the lower GAT decay heat curve into the ORNL simulation (along with the GAT pebble bed conductivity and specific heat relations) resulted in a total reduction of 100°C in the transient hot-node fuel temperature for the depressurized heatup. The ORNL decay heat

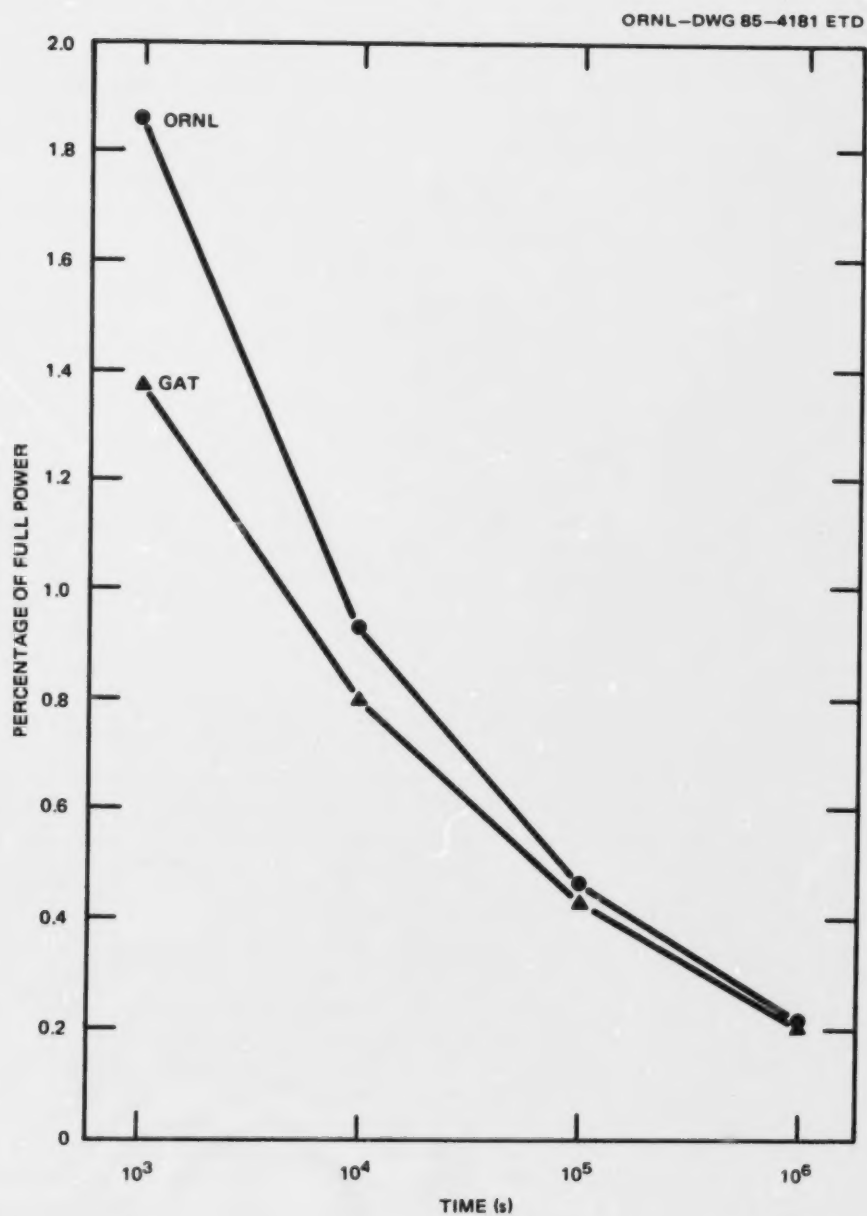


Fig. 5. Decay heat as a function of time since reactor scram.

correlation predicts the generation of a total of ~11% more decay heat during the ~30 h it takes to reach the transient peak.

The three input changes reported above would, if adopted into the ORNL models, bring the maximum predicted fuel temperature much closer to the 1600°C design goal for the depressurized heatup. It is not planned to adopt all three of these changes into the reference ORNL model because there is presently not sufficient justification for doing so; however, the linear fuel-bed thermal conductivity relationship presently in use by ORNL should probably be replaced with a power law curve similar to the GAT curve or to the KFA curve discussed in the next subsection. The power law correlations seem better able to provide for the increasing contribution of radiant heat exchange between pebbles as temperature increases above ~1400°C.

1.1.4 Effect of use of the KFA correlation for fuel conductivity

The KFA correlation⁷ for pebble bed thermal conductivity is similar to the GAT correlation, as shown in Fig. 4. It is significantly lower than the ORNL linear relationship in the lower temperature range and is only slightly above the ORNL conductivity in the high temperature range. Substitution of the KFA correlation into the ORNL simulation decreased the transient peak, hot-node fuel temperature by 8°C but increased the transient peak, average fuel temperature by 13°C. This result is reasonable because the KFA conductivity is lower in the range of the average fuel temperature during the transient but is higher in the range of the maximum core temperature.

1.1.5 Effect of modified side-reflector conductivity calculation

In a depressurized heatup the primary mechanism for heat removal from the core is conduction through the side reflector. Therefore, the thermal conductivity of the 100-cm-thick graphite side reflector would be expected to have some effect on the peak temperatures reached in a depressurized heatup. The initial ORNL simulation¹ utilized the conductivity vs temperature relationship illustrated by Curve 1 in Fig. 6. This correlation is based on Figs. 3-7 of the GAT Licensing Topical Report No. 13.⁴ Recent data published by GAT (the OG-3 tests⁸) for near-isotropic graphites such as H-451 show significantly higher thermal conductivity for both irradiated and unirradiated specimens. The data show that fast neutron ($E \geq 0.18$ MeV) irradiation has a significant effect on the thermal conductivity of the graphite.

To take full advantage of the OG-3 test results, two conductivity vs temperature relationships were used: one for the inner-half of the reflector (assumed to be fully irradiated) and the other for the outer-half of the reflector (assumed to be essentially unirradiated). Since the 50-cm-thick inner part is about three times the thickness of graphite required (on the average) to slow down a fission neutron, the outer part, quite reasonably, would not receive much fast neutron irradiation. Calculations published in support of the AVR design⁹ predict that the fast flux in the outer region would be lower than the fast flux in the inner region by a factor of at least 100.

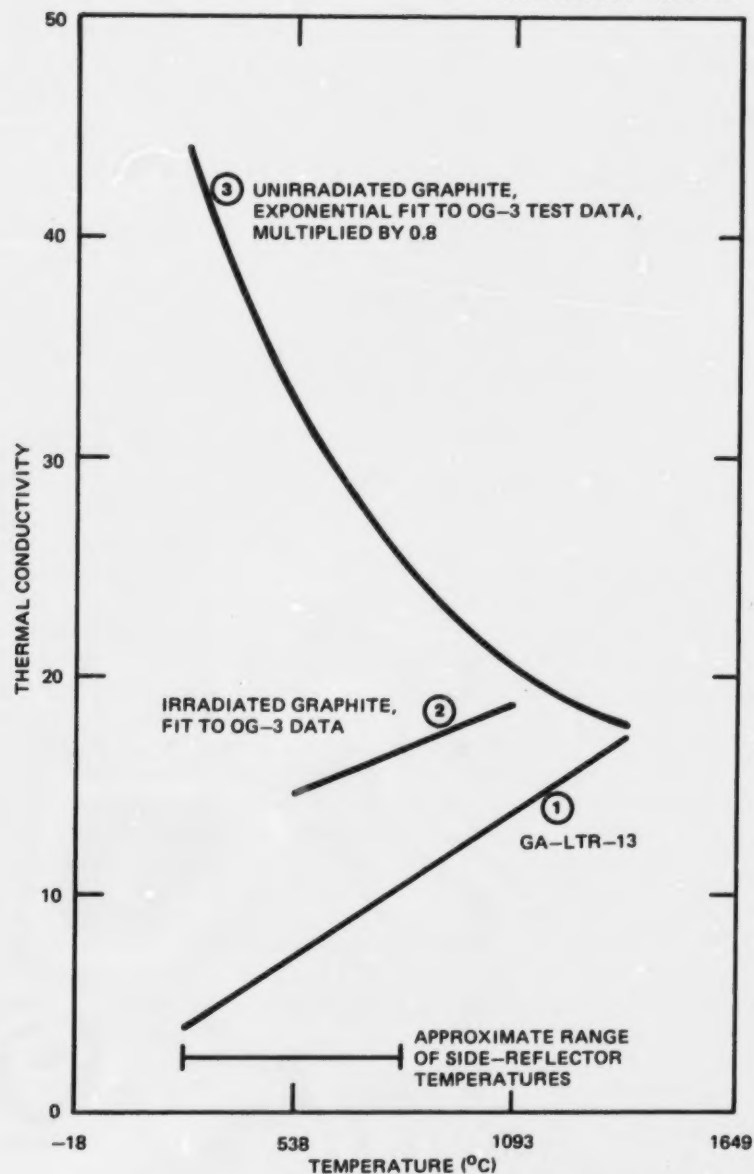


Fig. 6. Thermal conductivity of pebble bed side reflector.

Curve 2 of Fig. 6 shows the conductivity relationship assumed for the inner part of the reflector, and Curve 3 shows that used for the outer part. At the temperatures expected for each part of the reflector, the new conductivity for the inner 50 cm of the reflector (Curve 2) is about 50% higher than that previously assumed (Curve 1), and conductivity of the outer 50 cm is about 300% higher.

The new correlations for side-reflector thermal conductivity were programmed into the ORNL thermal model, and the depressurized heatup calculation was repeated. The results show that the assumption of higher conductivity for the side reflector reduces the transient peak, hot-node fuel temperature by 130°C.

1.1.6 Conclusions: sensitivity to modeling assumptions and input data

The results discussed above show that the transient peak fuel temperatures are sensitive to changes in decay heat, pebble bed conductivity and specific heat, and side-reflector conductivity. To establish with confidence the safety margins for the constantly evolving modular HTGR concept, further research will be needed to establish the applicability of the existing data base.

1.1.7 Modular HTGR heatup with loss of cavity-wall cooling

Studies described in the previous sections covered the depressurized heatup cases where the cooling system for the cavity cooling wall remained operative, either by forced convection cooling or by passive (boiling) cooling, following a depressurization and loss-of-forced-convection accident. This section describes a study of the longer-term reactor heatup problem that would evolve if the cavity-wall cooling is lost and if the ultimate heat sink is the earth surrounding the reactor vessel cavity. Initially, questions were raised about how good this heat sink would be, since some rough steady state heat conduction calculations showed that the reactor and vessel would have to reach extremely high temperatures before dissipating as much heat as was being generated by afterheat. Thus, calculations were in order that accounted for the large heat capacity of the surrounding earth, which (from simple observations about the small seasonal temperature variations of the earth at only modest depths) is known intuitively to be very effective at storing heat. Dynamic solutions to this cylindrical geometry heat conduction problem were generated using the IBM CSMP language's array integration feature that enabled, with little effort, a relatively fine-structure solution for the earth temperature history.

The total system model consists of a 2-node approximation of the reactor (average core and average side reflector), 1 node each for the reactor vessel and cavity concrete wall, and 50 radial (cylindrical-shell shaped) earth nodes. The parameters used for the effective heat capacity of, and conductance between, the core and side reflector were derived from the more detailed model described in Sect. 1.1 of Ref. 1. Where appropriate, conduction, convection, and radiation heat transfer are considered between components. Modeling of the concrete liner heatup optionally assumes either no cooling or a limited supply of passive (boiling water) cooling. The neglecting of axial conduction in the earth modeling is a conservative approximation. Soil properties vary quite a bit, depending on the sources of information and material assumptions, so a parameter study, along with earth node sizes, was done on these to indicate the sensitivity of the predictions to these ranges of uncertainties. The properties of various types of soils, along with

the data sources, are shown in Table 1. The outer radius chosen for the earth cylinder model ranged from 18.3 to 36.6 m for relatively short (500-h) transients to 107 m for the longer (3000-h to 1000-d) ones, the choice being a function of how far out the temperature perturbation penetrates. (The outer shell of the earth cylinder model is assumed to be insulated.) Since the results of this study differed markedly from the steady state assumption calculations, special care was taken in checking the individual internode heat flows and overall heat balances.

Table 1. Soil properties

Description	K (Btu/hr·ft·°F)	ρ (lb/ft ³)	C _p (Btu/lb·°F)
Dry soil ^a	0.2		0.44
Wet soil ^a	1.5		
Dry sand ^a	0.2	95	
Wet sand (10% moisture) ^a	0.6	100	
Limestone ^a	0.4	105	0.22
Stone concrete ^a	0.54	144	0.2
Coarse earth ^b	0.3	128	0.44
Earth's crust ^c	1.0		
Dry sand ^c	0.2		

^aFrank Kreith, "Principles of Heat Transfer," Second Edition, Scranton, Pa., International Textbook Company, 1965.

^bWarren Giedt, "Principles of Engineering Heat Transfer," D. Van Nostrand Company, Inc., Princeton, N.J., 1957.

^cBird, Stewart, & Lightfoot, "Transport Phenomena," John Wiley & Sons, Inc., 1960.

The results indicate that within the bounds of expected soil properties, afterheat can be absorbed in the surrounding earth without subjecting the core to extreme long-term temperatures, either when the cavity liner cooling system operates in the passive mode (with natural-circulation-fed, low-pressure boiling) for 2 weeks or when it fails immediately. In both cases, the maximum core temperatures are reached within the first day or two, and in this relatively short time the conduction to the surrounding earth is small, so the value of this peak temperature is independent of the earth modeling assumptions. For cases where the liner cooling system is operational, there is a secondary peaking of the average fuel temperature at 704 to 815°C after 2 to 4 months, depending on the assumed value of soil conductivity.

The predicted maximum reactor vessel temperature is very sensitive to the assumption of soil conductivity, where if a value in the higher range of k is assumed [~ 1.0 Btu/hr·ft·°F (173 W/m·k)], the maximum

temperatures are about 620 to 704°C. The higher temperature numbers in the range correspond to the cases where the cavity liner cooling system fails immediately. These peak vessel temperatures occur after 3 to 6 months, the time again depending on the soil modeling and liner cooling system availability assumptions.

1.2 Fission Product Release from HTGRs

J. H. Wilson R. L. Towns

The objective of this task is to generate experimental data required for the analysis of FP release in HTGR severe accidents. Initial efforts involve the determination of FP vapor pressures and diffusion rates through graphite. The experimental procedure consists of measuring the rate of loss at high temperatures from a mixture of powdered graphite and simulated FPs that has been placed in a 6.4-mm-OD x 3.2-mm-ID graphite tube. As the products diffuse through the tube wall, they are transported through a cold collection tube by argon carrier gas.

As described in previous reports, the experimental data have been analyzed assuming gas-phase bulk diffusion through the porous wall of the graphite tube. To determine whether the mode of transport is actually bulk diffusion or possible Knudsen diffusion, experiments were conducted under vacuum (1×10^{-3} atm) and at atmospheric pressure, using helium rather than argon as the carrier gas. Essentially no effect of pressure or carrier gas on the experimental diffusion rates was found, indicating Knudsen diffusion to be controlling (an effect of pressure that was reported last quarter was found to be a result of experimental problems). A sample of the graphite tube has been submitted for pore-size-distribution analysis to substantiate this finding.

Diffusion experiments utilizing the graphite tube were completed for Ru, La, Y, and Pr. As before, the sample was a mixture of powdered graphite and the particular elements. La, Y, and Pr were available as nitrates that were mixed with graphite and decomposed at high temperature before the diffusion experiments were performed. The vapor pressures, as determined using silver as a reference, were in all cases significantly different from the literature^{10,11} values reported for either the element or its carbide.

For verification of the results from the graphite tube tests, experiments had been performed in which a sample was placed in an impervious zirconia tube (that had an open end and a closed end) and the loss rate determined as a function of temperature. Here, transport occurs by diffusion through the gas space above the sample and not through the tube wall. Two data points were obtained with silver before the zirconia tube cracked. When tubes that have been ordered are received, the silver test will be completed, and data with palladium will be obtained.

1.3 Model and Code Development for Fission-Product Redistribution During Severe Accidents

C. F. Weber

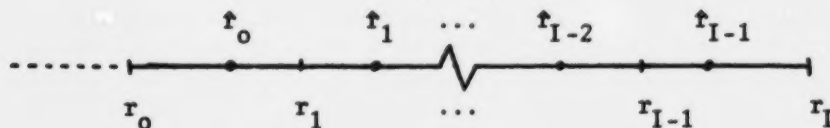
Investigation has continued in modeling and solution approaches for diffusion of nuclides through structural graphite. Recently, a program was written to calculate one-dimensional radial diffusion by using two different explicit finite-difference techniques. Both approaches have been verified on a sample problem with a known solution; however, one has demonstrated slightly, but consistently, better results.

Under either discrete method it is necessary to solve the following differential equations for C, the diffusing concentration of the FP group of interest, and B, the stationary concentration of the FP group of interest:

$$\frac{\partial C}{\partial t} = \frac{1}{r} \frac{\partial}{\partial r} \left(r D \frac{\partial C}{\partial r} \right) + \beta B - \gamma C \text{ and}$$

$$\frac{\partial B}{\partial t} = \gamma C - \beta B.$$

For numerical calculations the time variable is discretized as $t_k = k\Delta t$; the discrete radial mesh is of the form pictured below.



The first numerical approach involves only the r_i , which are chosen to be equally spaced so that $r_i = i\Delta r$. Using standard differences, the resulting explicit approximation to the first equation is:

$$C_{i,k+1} = C_{ik} + \frac{D}{r_i} \frac{\Delta t}{\Delta r^2} [r_{i+\frac{1}{2}} (C_{i+1,k} - C_{ik}) - r_{i-\frac{1}{2}} (C_{ik} - C_{i-1,k})] + \Delta t (\beta B_{ik} - \gamma C_{ik}).$$

The second approach uses both sets of discrete points, chosen to create mesh intervals of equal areas;

$$r_i^2 \equiv r_{i-1}^2 + \Delta A, \quad r_i^2 \equiv r_i^2 + \Delta A, \quad \Delta A \equiv (r_I^2 - r_0^2)/(I-1).$$

The first equation is discretized by integrating each term over the

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interval and approximating these integrals to get:

$$C_{i,k+1} = C_{ik} + \frac{\Delta t D}{\Delta A} \left[\frac{\hat{r}_i}{\Delta r_i} (C_{i+1,k} - C_{ik}) - \frac{\hat{r}_{i-1}}{\Delta r_{i-1}} (C_{ik} - C_{i-1,k}) \right] \\ + \Delta t (\beta B_{ik} - \gamma C_{ik}),$$

where $\Delta r_i = r_{i+1} - r_i$.

Although the first method is slightly faster, the second is more accurate; hence, the second approach will be used unless time constraints become important. The next step (already in progress) is to incorporate this diffusion calculation into the ORECA code itself.

1.4 Review of FSV Reactor Technical Specification on Limiting Maximum Core Temperatures

S. J. Ball

The ORNL review of the FSV tech spec limiting conditions for operation (LCO) 4.1.9, sponsored primarily by a subcontract with Nuclear Regulatory Commission (NRC)-Region IV, is part of a larger NRC program aimed at reviewing, clarifying, and (where feasible) simplifying the FSV tech specs.

In the course of the review, it was noted that some of the earlier misunderstandings about the scope and "completeness" of LCO 4.1.9 stemmed from the fact that by itself it does not limit plant operating conditions to prevent core overheating. Instead, it provides limits that prevent core flow stagnation and reversals, in which case the refueling region outlet thermocouples are to be relied upon to enable estimates to be made of maximum core temperatures. The ORNL recommendations generated in this quarter included fixes for a "gap" in the coverage of allowable outlet temperature readings, a proposed method for improving the "official" heat power estimate at low powers, and a method to improve low-range primary system flow estimates by using circulator speed and performance map data. Based on discussions with Public Service Co. of Colorado (PSC) and GAT, it was concluded that the current version of the ORECA code, which models each of the 37 refueling regions' flows with a single-channel approximation, would not suitably address the problems of flow stagnation since it does not account for intraregion flow non-uniformities. These nonuniformities are due primarily to "tilt" factors in the power densities, where within a region the ratio of maximum-to-average power may be as high as 1.6. Model and code development efforts for including these features in ORECA were begun.

1.5 Cooperative Programs with the
Federal Republic of Germany

J. C. Cleveland

No work was performed on this task during this quarter because the agreement for the cooperative program between the West Germany Ministry of the Interior and the NRC has not yet been formally approved.

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2. TRIP MADE UNDER PROGRAM SPONSORSHIP - MEETING AT
NRC REGION IV OFFICES ON FORT ST. VRAIN
REACTOR TECHNICAL SPECIFICATIONS

S. J. Ball

In conjunction with the task discussed in Sect. 1.4, S. J. Ball traveled to Arlington, Texas, to participate in a meeting between NRC and PSC representatives. The purpose of the meeting was to discuss the review of the Fort St. Vrain LCO 4.1.9 technical specification. S. J. Ball presented the preliminary ORNL findings; PSC representatives presented background information relating to LCO 4.1.9.

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